NON-PUBLIC?: N
ACCESSION #: 9012280132
LICENSEE EVENT REPORT (LER)

FACILITY NAME: Nine Mile Point Unit 1 PAGE: 1 OF 7

DOCKET NUMBER: 05000220

TITLE: Reactor Scram During Surveillance Test Due To Personnel Error
EVENT DATE: 11/17/90 LER #: 90-026-00 REPORT DATE: 12/17/90

OTHER FACILITIES INVOLVED: N/A DOCKET NO: 05000

OPERATING MODE: N POWER LEVEL: 096

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR SECTION:
50.73(a)(2)(iv)

LICENSEE CONTACT FOR THIS LER:
NAME: Robert Tessier, Manager TELEPHONE: (315) 349-2707
Operations Unit 1

COMPONENT FAILURE DESCRIPTION:
CAUSE: D SYSTEM: AA COMPONENT: LS MANUFACTURER: M040
REPORTABLE NPRDS: No

SUPPLEMENTAL REPORT EXPECTED: No

ABSTRACT:

On November 17, 1990, at 2109 with reactor power at 96 percent, Nine Mile Point Unit 1 (NMP1) experienced a full reactor scram, High Pressure Coolant Injection (HPCI) mode of feedwater initiation and Electromatic Relief Valve (ERV) initiation due to Main Steam Isolation Valve (MSIV) isolation. The MSIV isolation occurred as a result of an incorrect fuse being removed during a surveillance test of a Main Steam Line Radiation monitor. The plant was safely shutdown and no adverse safety consequences resulted.

The primary cause of this event was determined to be personnel error, specifically poor work practices.

Immediate corrective actions included placing the plant in a safe shutdown condition and resetting the reactor scram. Subsequent corrective actions included counseling and discipline of operations

personnel involved in the test performance.

END OF ABSTRACT

I. DESCRIPTION OF EVENT

On November 17, 1990, at 2109 with reactor power at 96 percent, Nine Mile Point Unit 1 experienced a full reactor scram, High Pressure Coolant Injection (HPCI) mode of feedwater initiation (low reactor water level) and Electromatic Relief Valve (ERV) initiation (reactor pressure > 1090 psig) due to Main Steam Isolation Valve (MSIV) isolation. The MSIV isolation occurred as a result of an incorrect fuse being removed during surveillance testing of a Main Steam Line (MSL) radiation monitor. At the time of the event the reactor mode switch was in RUN, the reactor coolant temperature was approximately 524 degrees Fahrenheit and reactor pressure was at 1021 psig. The following is a sequence of events leading up to the plant trip.

During the week of November 12, 1990, a determination was made that MSL Radiation Monitor 112 was drifting. The instrument was declared inoperable and per Technical Specification (T.S.), and the Monitor was placed in the tripped condition in the Reactor Protection System (RPS). Following this, the Instrument and Control Department (I&C) performed the maintenance necessary to correct the drifting problem. At the conclusion of maintenance work, Operations and I&C determined that Operations Surveillance Procedure N1-ST-W4, "Main Steam Line Radiation Monitor Instrument Channel Test" would be the appropriate means of proving operability. The procedure would be used as a Post Maintenance Test (PMT) as well as meet the requirements for Operations weekly surveillance on the Radiation Monitors.

On the afternoon of November 16, 1990, a determination had been made by Licensing and Operations that surveillance testing of certain instrument channels was not being conducted in accordance with Technical Specification requirements (see LER 90-24). A management decision was made to revise surveillance procedures to require that certain instrument channels be placed in the tripped condition before performing surveillance testing. This included the MSL Radiation Monitors test N1-ST-W4.

This required Operations personnel to perform an extensive Temporary Change Notice (TCN) to N1-ST-W4 to incorporate the methodology that would: (1) place each Radiation Monitor channel in the tripped condition prior to performing the channel test, and (2) permit verification of the necessary alarms and trips associated with each channel. This TCN was prepared by the day shift Station Shift Supervisor (SSS) on Saturday, November 17, 1990, and reviewed by the evening shift SSS and his assistant with

I. DESCRIPTION OF EVENT (cont.)

the day SSS. At 20:59 (Saturday evening), the Nuclear Auxiliary Operator E (NAOE), was given permission to test MSL Radiation Monitor 112 using N1-ST-W4. Twenty-eight steps were completed as required. During one of those steps a MSL Hi Radiation trip signal, which the NAOE had put into the MSL Radiation Monitor 112, automatically reset. The MSL Hi Radiation trip signal is designed to reset within three minutes. This then required him to re-insert the trip signal before he could continue. This trip signal produces a half scram and a 1/2 MSIV signal. After confirming the associated alarms and trips, he proceeded to manually insert Channel 12 trips before removing the MSL Hi Radiation trip signal at the monitor. This required pulling two fuses, one associated with the scram logic and one associated with the isolation logic. After pulling the proper scram logic fuse, he proceeded to pull the isolation logic fuse. At 21:09 step 8.3.29 was performed and states "Pull Sensor Relay Output Logic F-53 in M panel (yellow label)". When the NAOE entered the M panel in the control room, he identified F-53 without regard to color code. Subsequently, when he pulled the fuse (which was actually F-53 green label in RPS Channel 11), an RPS isolation signal was generated completing the logic for a full MSIV isolation and resulting in a reactor scram.

Reactor water level dropped to +15 inches, resulting in HPCI initiation, which ultimately restored water level to approximately +100 inches. Turbine/Generator trips came in as expected, however, Power Board 11 failed to fast transfer to reserve power. Operators immediately restored the Power Board. The scram was reset at 21:32. At 00:15 the Nuclear Regulatory Commission (NRC) was notified by telephone of the event in accordance with 10CFR50.72 (b) (2) "4 hour non-emergency notification".

There were no inoperable systems which contributed to this event. However, this event resulted in several component failures. As mentioned above, Power Board 11 did not transfer to reserve automatically and had to be closed manually. This momentary loss resulted in tripping of Recirculation Pumps 11 and 12 and momentary loss of HPCI train #11 components (Condensate Pump 11, Feedwater Booster Pump 11, and Feedwater Pump 11). However, the HPCI train #11 components immediately restarted when power was restored to Power Board 11. Scram Discharge Volume (SDV)

High Level Scram Switch 44.2-35A was supposed to reset within 20 sec nds,
following resetting of a scram, but did not reset until 15 minutes later. The effect of the SDV Level Switches' failure to reset was negligible since the SDV High Level Scram Switches are bypassed

I. DESCRIPTION OF EVENT (cont.)

following a scram. Two minor problems with the Control Rods were also experienced. Control Rod 06-35 had missing position indication from position 46 through 30 (identified on the 30 channel brush recorder). Also Control Rod 22-11 traveled to its full position but settled to position 02. This is an identified problem with GE BWR's where the rod actually reaches 00 then rebounds to 02. NMP1 has analyzed for all rods scramming to position 02.

II. CAUSE OF EVENT

A root cause analysis was performed using Nuclear Division Procedure 16.01 and the root cause was determined to be personnel error, specifically poor work practices. The operator did not self-verify the correct component before action was taken. This resulted in the wrong fuse being pulled, despite instructions in the procedure which identified (by color) the fuse to pull.

However, contributing causes, which increased the probability of personnel error, included:

1. Managerial methods: Management decision to make extensive changes through a TCN instead of a formal procedure review and re-typing; Also, SSS's decision to use one person to perform the surveillance procedure requiring time critical actions at multiple locations.

2. Man-machine interface: RPS fuses in M panel were not specifically identified by RPS channel and sub-channel numbers (insufficient labeling). See discussion under Additional Information concerning ongoing corrective actions in this area.

3. Training: Operator training does not include RPS panel walkdown and color code designation instruction.

4. Written procedure and documents: TCN'ed procedure did not place appropriate emphasis on steps/information. When the TCN was written the step to pull the fuse did not identify the fuse by RPS channel

and subchannel numbers. In addition, information enclosed in parenthesis is generally included for enhancement, not the most important piece of information in the statement.

III. ANALYSIS OF EVENT

This event is reportable in accordance with 10CFR50.73 (a)(2)(iv), "Any event or condition that results in manual or automatic actuation of any Engineered Safety Feature (ESF), including the Reactor Protection System (RPS). However, actuation of an ESF, including RPS, that results from and was part of the preplanned sequence during testing or reactor operation need not be reported".

There were no significant safety consequences as a result of this event nor was the reactor in an unsafe condition at any time. The initial unavailability of HPCI train 11 due to loss of Power Board 11 was of minor consequence as HPCI train 12 met the necessary reactor vessel make-up requirements. As previously mentioned, Control Rod 22-11 traveled to its fully inserted position but settled to position 02. This is an identified problem with GE BWR's where the rod actually reaches 00 then rebounds to 02. NMPC has analyzed for all rods scramming to position 02.

This event is similar and bounded by the "Main Steam Isolation Valve isolation with scram" event discussed in Final Safety Analysis Report, Chapter 15, Section 3.5.

IV. CORRECTIVE ACTIONS

Subsequent corrective actions include:

1. The personnel involved have been counseled and disciplined concerning their actions on November 17, 1990. Special emphasis was placed on ensuring that procedures involving new or unusual operator actions are adequately walked down and reviewed. Also emphasized was the importance of self-checking to ensure correct component identification before taking actions.

2. Develop and implement a training module for plant operators to include the general layout inside control room panels and an explanation of the existing color coding system for RPS bus circuitry.

3. The TCN was re-written (and formally typed) after incorporation of

operating shift comments and a human factors review. N1-ST-W4 was successfully performed during subsequent testing of the MSL Radiation monitors. Longer term corrective action involves approval of Administrative Procedure AP-2.2,

IV. CORRECTIVE ACTION (cont.)

"Procedure Change Notices and Procedure Evaluation Requests". This procedure will contain specific criteria for TCN's.

4. A Work Request (WR) was written to have I&C troubleshoot missing position indication on control rod 06-35.

5. The most probable cause of the failure of Power Board 11 to transfer is improper position of contacts within the SB control switches for breakers R113 (normal supply) and R112 (reserve supply) in the closing circuit for breaker R112. A formal root cause has been requested to investigate the problem.

6. Regarding the slow reset of SDV level switches, procedures are being developed to calibrate the switches and a program is being developed to cyclically calibrate the SDV level switches.

V. ADDITIONAL INFORMATION

A. Previous similar events:

LER 84-17 "Reactor Scram During Surveillance Test". In LER 84-17, an MSIV isolation signal and subsequent scram resulted from the removal of a fuse as part of surveillance test N1-ST-V8. In that event, deficiencies in the surveillance procedure were credited with causing the event and N1-ST-V8 was corrected to prevent recurrence. The corrective actions taken by this LER would not have prevented the events leading to LER 90-26.

LER 89-11 "Automatic Initiation of Reactor Building Emergency Ventilation Due to Poor Work Practice". The event discussed in this report is similar to the events discussed in LER 89-11. In LER 89-11, an ESF actuation (Reactor Building Emergency Ventilation) occurred. The root cause (as in the event discussed in this LER) was poor work practices. Similar contributing factors included: (1) Inattentiveness to detail and failure to research the plant impact of pulling process radiation fuses, and (2) Lack of unique circuit designation in regards to the labeling of fuses in RPS Bus 11 and 12 Power Supply Cabinets. The final corrective action for

V. ADDITIONAL INFORMATION (cont.)

that event was to develop a plan to upgrade fuse designations for all safety related equipment by February 1, 1990, and to have this implemented as part of a design basis reconstitution program in 1991. Once the fuse designation upgrade program has been implemented, the probability of repeating the events that lead to LER's 89-11 and 90-26 will be greatly reduced.

B. Identification of components referred to in this LER:

IEEE 803 IEEE 805
COMPONENT FUNCTION SYSTEM

High Pressure Coolant Injection NA BJ
Main Steam Isolation Valve ISV SB
Electromatic Relief Valve RV BF
Power Board #11 CB EB
SB Control Switches for Breakers IC EB
Control Rod NA AA
Scram Discharge Volume Level Switches LCO AA

C. Failed components:

Power Board 11 - General Electric
SDV Level Switch - Magnetrol
Control Rod 06-35 - General Electric

ATTACHMENT 1 TO 9012280132 PAGE 1 OF 1

NIAGARA
MOHAWK

NINE MILE POINT NUCLEAR STATION/P.O. BOX 32,
LYCOMING, N. Y. 13093/TELEPHONE (315) 343-2110

NMP73973

December 17, 1990

United States Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555

RE: Docket No. 50-220
LER 90-26

Gentlemen:

In accordance with 10CFR50.73, we hereby submit the following Licensee Event Report.

LER 90-26 Which is being submitted in accordance with 10CFR50.73(a)(2)(iv), "Any event or condition that resulted in manual or automatic actuation of any Engineered Safety Feature (ESF), including the Reactor Protection System (RPS). However, actuation of an ESF, including the RPS, that resulted from and was part of the preplanned sequence during testing or reactor operation need not be reported".

A 10CFR50.72 report was made 0015 hours on November 8, 1990.

This report was completed in the format designated in NUREG-1022, Supplement 2, dated September 1985.

Very truly yours,

Joseph F. Firlit
Vice President - Nuclear Generation

JFF/DPS/lmc

ATTACHMENT

cc: Thomas T. Martin, Regional Administrator Region I
W. A. Cook, Sr. Resident Inspector

*** END OF DOCUMENT ***